



**Third United Nations
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Abstract

THE IRRADIATION BEHAVIOUR OF GRAPHITE

by

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*B. T. Kelly

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*W. N. Reynolds

Experimental results on the effects of irradiation on graphite in the temperature range 150-650°C are described. Samples of Grade A reactor graphite have been irradiated in high flux reactors and measurements made of the changes in thermal conductivity, thermal expansion, Young's modulus and dimensions. The changes in thermal expansion and dimensions of pyrolytic graphite were also measured. As far as these properties are concerned, it is considered that the results give a good representation of the behaviour of the crystals of which reactor graphite is composed and the dimensional changes of pyrolytic graphite are compared with values for the dimensional changes of graphite crystals, inferred from measurements on reactor graphite.

The results of recent work on the structure of radiation defects in graphite are also summarised. Thin film electron microscopy has yielded a considerable amount of direct information on the properties of large interstitial clusters and some of the properties of vacancies and small interstitial clusters can be inferred. The results of the irradiation experiments are interpreted in terms of a radiation damage model, formulated on the basis of the new information on defect structures.

This work extends the data on the behaviour of reactor graphite over a sufficiently wide range of temperature and dose to cover almost the complete life of the reactors in the first stage of the United Kingdom's power programme, and improved understanding of the mechanisms involved has enabled methods of allowing for differences in temperature and flux level to be developed.

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Abstract

URANIUM AND THORIUM ORE RESERVES AND POTENTIAL

* S.H.U. Bowie

The confusion that exists concerning presently estimated reserves of uranium and thorium results largely from the loose usage of the term 'reserves'. Therefore, the terms 'reserves', 'potential' and 'resources' are defined and an indication given of the degree of confidence that can be placed in published or reserved data.

A brief account is given of the history of the discovery of the major uranium fields and of the build-up of ore reserves during the period 1945-55. Reserve data are discussed and comments made on the potential of the known fields as well as of the possibility of finding additional provinces with comparable tonnages of ore. It is noted that no major ore deposit has been found since 1955 and that uranium reserves at the end of 1961 have been revised downwards by more than 30% as compared with figures for the end of 1958. This has resulted because ore reserves were estimated mainly from surface and borehole data which gave an inadequate picture of changes occurring in depth.

Of equal importance to reserves, from the long-term point of view, is the availability of ore. It is noted that if no major discoveries of uranium are made and current production maintained, by 1975 the reserves of the U.S.A. will be largely depleted and output in subsequent years can be expected mainly from the Republic of South Africa and from Canada. However, production from the former country is not likely to exceed 6000-7000 tons U_3O_8 a year and the output from the latter may not bear a normal relationship

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to her total reserves, since approximately half the known uranium ore is in one property.

Comments are made on the changes in prospecting technique that will be necessary to locate hidden deposits of uranium and thorium. Also, an indication is given of the basic research required to minimize the cost of exploration for additional ore when this becomes essential. Reserves of thorium are not known with the same degree of certainty as for uranium, but tentative figures indicate that they are of the same order as for uranium.



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Abstract

THERMAL REACTOR PHYSICS STUDIES ON PLUTONIUM FUELS

by

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Results of experimental measurements on a range of plutonium-fuelled, water and graphite moderated subcritical and critical assemblies are described and compared with calculations using multigroup diffusion and transport theory methods.

Brief accounts are given of the equipment, fuel elements and experimental techniques. Bucklings of assemblies fuelled with thin plutonium aluminium alloy plates, and with plutonium-uranium alloy bars have been determined by the exponential method. Relative reaction rates of various nuclides have been measured in these assemblies to provide a check on accuracy of spectrum calculations.

Experiments on a plutonium-fuelled graphite moderated critical assembly have been carried out to obtain temperature coefficients of reactivity and power distributions. A direct measurement of the neutron energy spectrum in this system has been made by the time-of-flight technique.

A new 40-group scheme of calculation based on the Winfrith DSN code is described. A discussion of the basic nuclear data for fuel and moderator used in this scheme is given.

The agreement achieved between theoretical prediction and experimentally determined reactivity, temperature coefficients, reaction rates and neutron energy spectra is discussed, and compared with that found for similar uranium-fuelled systems.

Conclusions are presented concerning the adequacy of the theoretical methods and nuclear data, and proposed extensions of the theoretical and experimental work to further systems is described.

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Abstract

FAST REACTOR PHYSICS, INCLUDING RESULTS FROM U.K. ZERO POWER REACTORS

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* A.R. Baker * J.L. Rowlands
ø J. Weale £ J.E. Mann
+ D.C.G. Smith * E.P. Hicks

The nuclear data needed for fast reactor calculations covers a wide energy range and present measurements are not sufficiently comprehensive and accurate to be used as the sole basis for calculation. Thus the calculations for fast reactors commonly use sets of data in a number of energy groups in which the contents have been selected or adjusted as a result of measurements on zero power reactors. The sets used are reviewed and their defects discussed. A new approach using GALAXY and the Data File has been developed in the U.K. which approaches the ideal of calculations based solely on basic nuclear data.

Calculations for fast reactors use a number of well-proved transport and diffusion theory codes, which can be used with up to about 30 groups in one or two dimensions. These calculations give the spectrum above about 50 Kev fairly well. Problems arise in extending these calculations to three dimensions and in covering the region from 50 KeV to 50 eV where resonance effects are

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ø U.K. Atomic Energy Authority, A.W.R.E. Aldermaston
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important. This energy region is of vital interest in the calculation of sodium and Doppler temperature and power coefficients.

The dynamic behaviour of the reactor and the control system used depends on the values of these power and temperature coefficients, though it is possible to design an entirely satisfactory control system even with unfavourable power coefficients.

A more stringent limitation on possible values of temperature coefficients arises in the study of reactor accidents. The types of accident that need to be studied are reviewed and their interpretation with the reactivity coefficients considered. These accidents include coolant flow failure, accidental ramp or step reactivity additions. The reliability of present methods of calculating the dynamic behaviour of a reactor is considered in the light of the agreement between calculation and experiment on the D.F.R. These methods are used to predict the behaviour of a large power fast reactor, and the importance of the Doppler temperature coefficient of reactivity demonstrated.

The Doppler effect also has a profound influence in the case of the hypothetical accident resulting from the melt-down of a fast reactor core and leading to the formation of a super-prompt critical assembly. It is shown that a relatively small Doppler coefficient will reduce the energy yield in the resulting power excursion by a large factor.

In order to fix the design of a power reactor it is necessary to make a choice out of a large number of variable design parameters. To make a systematic choice possible a computer programme has been written to enable the effects of variation in a basic design to be assessed rapidly. The capabilities of this programme are discussed and some results of the survey are presented.

The data and calculation methods used in the design of the P.F.R. are being checked against measurements made in zero-power assemblies. Results from Vera, Zebra and Quagga are quoted and the choice of assemblies to emphasise particular nuclear data discussed. The experimental results are then compared with the theoretical predictions for critical mass, spectrum and reaction rates and the observed discrepancies discussed. The prediction of the Doppler coefficient of a power reactor can be improved by measurements made in a zero power reactor - and several experiments are planned.



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Abstract

NEUTRON INTERACTIONS WITH REACTOR MATERIALS

by

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* P. A. Egelstaff

* A. T. G. Ferguson

The UKAEA programme of measurements of neutron interactions with fissile, fertile, moderator, coolant and structural materials is briefly discussed in the light of reactor physics demands. The reasons behind the measurements and the extent of the programme are summarised in the first section while the remainder of the paper is devoted to the description of the experimental work on typical examples of each of the categories mentioned, namely uranium-235, uranium-238, iron and water.

The section on uranium-235 describes measurements of the fission cross section in various neutron energy regions from thermal to 14 MeV. The variation of the quantities $\bar{\nu}$ and η with neutron energy is presented, the former over a wide energy range and the latter in the resonance region. A determination of the total cross section up to 80 KeV is also given, from which s and p-wave strength functions and the absorption cross section are deduced.

The work on ^{238}U is illustrated by studies of the capture cross section throughout the energy range from a few eV to 14 MeV and of inelastic scattering from threshold to 4 MeV. Recent improvements in technique permit the construction of much more complete energy level schemes for the ^{238}U nucleus.

Nuclear inelastic scattering in iron is also described in the energy range 1 to 4 MeV and optical model fits are given to the data. Cross sections and spectra for the emission of γ -rays following inelastic scattering are described,

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and the capture cross section below 100 KeV is briefly discussed.

The final section of the paper deals with the scattering of thermal neutrons by bound nuclei. The results of measurements do not contain the complete information required for reactor calculations and must be supplemented by theoretical considerations based on suitably chosen models of the moderator materials. The data for water at different temperatures are presented and compared with several theoretical models.



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Abstract

EVALUATION, STORAGE AND PROCESSING OF NUCLEAR DATA
FOR REACTOR COMPUTATIONS

by

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* M. F. James * I. C. Pull
ø W. M. M. Kerr + P. Schofield

Through the use of large computers reactor calculations are nowadays much more precise and detailed than was possible in the early years of the atomic energy industry. This advance is being supported by large numbers of neutron cross-section measurements of improved accuracy and extent. However preparation of the derived quantities needed for reactor calculations can be extremely laborious. To surmount this difficulty in a systematic way workers in the UKAEA are building up a data library on punched cards and magnetic tape of the complete range of basic nuclear cross-sections which are needed, with versatile processing programmes to generate the derived quantities.

Part 1 of the paper describes this work. The process of evaluation of cross-sections for the library is discussed. For a library to be useful in reactor calculations it must contain a complete range of data for the most important materials. Thus the initial stage of its construction is quite a large undertaking: it is now complete so far as the UKAEA data library is concerned. At the present time this contains neutron cross-section data in the range 1 meV to 15 MeV for 39

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materials, including angular and energy distributions of secondary neutrons. Some gamma ray cross-sections in the range 10 keV to 20 MeV are also available.

The format of the library is very flexible, and both tabular and parametric representations of the data are allowed. The latter have considerable advantages in some parts of the energy range if computer programmes are available for their interpretation. Some such programmes are in use and others are being developed. Because the data are very numerous several ancillary programmes are used to mechanise jobs such as checking, altering and printing the library.

The principal "processing programmes" developed for general use with the library are GALAXY, which generates multigroup cross-sections, and DICE which prepares the data for Monte Carlo calculations and aids consultation of the prepared data.

To extend the library and keep it up-to-date in the light of new measurements is a continuing task which is likely to become more exacting and urgent as the output of new experimental data increases. International collaboration is needed, to obtain full value from the experimental work. Some informal collaboration has already been started.

Part 2 summarises work on the evaluation of data on the thermal neutron scattering law; several of the points made in general terms in part 1 are aptly illustrated by this example. The differential scattering cross-section cannot be measured over the whole range of initial and final energies and scattering angles at all the temperatures required. So it is necessary to make considerable use of theory in interpreting and extrapolating the measurements. The adequacy of the scattering laws obtained in this way are checked by appeal to measurements of neutron spectra in moderating assemblies.



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Abstract

CALCULATION OF REACTOR POWER AND TEMPERATURE
DISTRIBUTION IN THREE SPATIAL DIMENSIONS

by

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+J. J. Syrett

°J. G. Tyrer

xDr. J. Walsh

=Dr. H. E. Wrigley

The paper describes the development of a large computer program written for the IBM 7090 computer to calculate steady state reactor power and temperature distributions in three spatial dimensions.

The major part of the calculation is the solution of the group diffusion equations for up to five energy groups in a non-linear form, i.e. with explicit reactivity feed-back effects due to material temperatures and to xenon-135 concentration. The diffusion equations are solved by finite-difference methods using a point successive over-relaxation iterative scheme with Chebyshev extrapolation of the sources. The ancillary equations defining temperatures and xenon-135 concentration are solved periodically; the coefficients of the diffusion equations are then re-evaluated and the finite-difference matrices modified accordingly before proceeding with further flux iterations. The reasons for the choice of numerical techniques employed are discussed and programming methods and problems are noted.

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=Atomic Energy Establishment, U.K.A.E.A., Winfrith

Two new techniques have been developed to increase the scope of the diffusion calculation. These are firstly, a method of representing discrete absorbers (control rods) at any point within a finite difference mesh and secondly, a method of super-imposing a fine structure calculation of the source-sink type on the macroscopic flux distribution obtained from the diffusion theory calculation. These techniques (which are described in some detail) have been developed in order that detailed studies of complete power reactors can be made without the use of an excessive number of finite difference mesh points when solving the diffusion equation.

At the time of writing (December 1963), the main diffusion program together with subsidiary programs to calculate flux fine structure and absorber data have all been written and tested. The diffusion program has shown satisfactory rates of convergence in both two group and five group cases and the computation time is considered to be reasonable.

Intended future use of the program by both the C.E.G.B. and the U.K.A.E.A. is discussed and the extension of the basic steady state diffusion program to a time-dependent model for the study of reactor transients is also noted.



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Abstract

EXPERIMENTAL BASIS OF POWER REACTOR OPERATION

by

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+D. J. Millard
+R. Russell

Current methods for obtaining essential operational data on power reactors are almost entirely empirical or semi-empirical in nature because of the present inadequacies of purely theoretical methods of physics calculation. Consequently it has been necessary on the first U.K. civil power reactors at Bradwell and Berkeley to repeat many of the measurements performed on the U.K.A.E.A. reactors at Calder and Chapelcross. The main purpose of all the measurements is to provide data which can be used to define the method of operation which achieves the highest efficiency and output consistent with the assurance that the reactor can be controlled and shut down in all credible fault conditions without the safety margin on any component being exceeded. The paper is mainly confined to a discussion of parameters which determine the net reactivity of these reactors, and to the relevance of kinetic experiments.

The assessment of control rod calibration as a function of irradiation and temperature provides a scale for measuring other reactivity parameters, such as the change in reactivity due to the formation of plutonium and fission products, the total temperature poisoning arising from fuel and moderator temperature coefficients of reactivity and reactivity taken up by absorbers. A comparison of the results with theoretical calculations leads to an assessment of the error in applying such methods to the prediction of operational data concerning such problems as fuel cycles and shut-down capacity. In general it is shown that the combination of experiment and theory can satisfy operational requirements within the necessary limits.

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+ Central Electricity Generating Board, London

Kinetic experiments are described which have satisfied four main objectives: to confirm the validity of theoretical models used to calculate the consequences of certain credible fault conditions, from which operational limits on such parameters as fuel element temperature and shut-down capacity under fault conditions can be defined; to measure the response of reactors to small perturbations likely to arise in practise; to confirm the model used for stability studies; to calibrate control rods.



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Abstract

**EXPERIENCE IN RELATING THE OBSERVED THERMAL PERFORMANCE
OF CIVIL MAGNOX-URANIUM POWER REACTORS TO DESIGN
PREDICTIONS AND OPERATING CRITERIA**

by

* P.R.J. French
+ D.J. Millard
* M.J. Cartmell
* B.K. Chesterton

1. The paper is based on experience gained from the operation of the nuclear power stations at Berkeley, Bradwell, and Latina, and is concerned with the relationships between the power output of the reactors, the distributions over the reactor core of gas temperature and fuel element temperature, and the temperature limitations adopted in design and operation.

2. The derivation of the temperature limitations that governed the initial design of the reactors is outlined, and the instrumental facilities that are available for checking that these limitations are met during operation are summarised. It is shown that the measurements of fuel element temperature and fuel channel gas outlet temperature are made on the basis of sampling the actual distributions over the very large numbers of elements and channels.

3. During the early operation of the stations, the gas outlet temperature and the power output were restricted by criteria which

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+ Central Electricity Generating Board, Berkeley Power Station

were based on the predicted behaviour of the reactors under fault conditions. The paper explains the derivation of these criteria, and of a rapid approximate method of temperature assessment which enabled the fault criteria to be translated into simple Operating Rules concerning the maximum permitted values of the indicated gas and fuel element temperatures. Technical progress has since enabled the operating restrictions to be relaxed, so enabling all the stations to be operated at design output or above. The main contributions towards this relaxation have been the satisfactory experience of early operation, improvements to the protective equipment of the reactors, out-of-pile research into the behaviour of the fuel elements under fault conditions, and improvements in the techniques used for the statistical analysis of the operational temperature measurements.

4. A description is given of a method of analysing the operational temperature measurements which has been programmed for a digital computer, and has been found to be convenient and effective in practice. The method yields print-outs in the form of core plans showing the inferred systematic distributions, channel by channel, of fuel element temperature and channel gas outlet temperature, under both steady-state and fault transient conditions. The analysis also determines the standard deviations of fuel element temperature and gas temperature, so that estimates of the random maximum temperatures under normal and fault conditions can be made.

5. Methods available for the prediction of temperature distributions for given operating conditions are outlined, and predictions made by these methods are compared with observations at the three stations.

6. A table is included to show the comparison between the design power output and temperatures, the conditions existing during early operation and more recent conditions at all the stations.



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Abstract

STUDIES OF RESONANCE ABSORPTION AND THE DOPPLER
PHENOMENON FOR FAST AND THERMAL REACTORS

by

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Miss J. E. Beardwood

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Miss H. M. Sumner

U.K. Atomic Energy Authority, A.E.E., Winfrith

Part I of the paper describes some investigations of resonance absorption and its temperature dependence in dilute fast reactors. Two particular problems are discussed with special reference to the Doppler temperature coefficient:- the overlapping of resonances of the fissile and fertile constituents of the fuel, and the consequences of using multilevel resonance theory instead of the customary single level theory. A new numerical method for treating the resonance energy region of dilute fast reactors is then described. This method allows automatically for resonance overlapping and multilevel theory as well as for core heterogeneity, and enables the detailed neutron energy spectrum between 20 kev and thermal energies to be calculated. Some results from a preliminary test of the method for a homogeneous mixture of uranium 238 and carbon are compared with calculations using a conventional method.

Part II deals with the calculations of resonance integrals for heterogeneous thermal reactors. The use of equivalence theorems, which enable results for a homogeneous medium to be applied to a heterogeneous system, is explained. An improved form of the standard equivalence theorem for isolated solid fuel rods is derived. The application of equivalence theorems to treat regular arrays or finite clusters of fuel rods is discussed, introducing the concept of 'effective surface'. A method of calculating this parameter for liquid-filled arrays or clusters is outlined. Calculated values

of the resonance integral and its temperature dependence are then compared with experimental data on UO_2 rods and clusters, and also with results obtained by the Monte Carlo method.

An investigation is made of the resonance integral for fissile isotopes when mixed with uranium 238, with particular attention to the effect of resonance overlapping. Numerical results are compared with measurements of the relative reaction rates of fissile samples in uranium and copper bars, which were irradiated in a uranium-fuelled graphite lattice.



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Abstract

THE PHYSICS OF THE DOUNREAY FAST REACTOR AT POWER

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- * K.W. Brindley
- * D. Tait
- * R.C. Wheeler

The physics of the Dounreay Fast Reactor has been the subject of several papers which have covered the period from first criticality to operation at 11 MW. The present paper deals with physics problems of the reactor at power and extends the published work to the end of 1963.

A. Statics

A brief summary of recent modifications to D.F.R. is given. By comparing several calculations, using various computer codes and cross-section libraries, the critical mass of D.F.R. may be estimated to 1% in spite of its complex core. An assessment of the required accuracy is given. Measurement of reactor power and fuel rating is made in several ways which are compared and differences explained. The ^{238}U (n, γ) reaction rate in the radial blanket has been determined by radio-chemical analysis of irradiated bulk uranium (breeder elements) and small samples of powder. Agreement with calculation is within experimental error (10%). A significant improvement has been made in agreement between measured and calculated reaction rates near the core boundary by adopting a hexagonal model for the calculation. In order to increase the fission rate obtainable in experimental rigs in the radial blanket the effect of replacing the uranium by a number of materials was investigated. Replacement by steel has increased the ^{235}U fission rate by 15-25% and an additional gain of 1.3% reactivity was achieved. By experiments involving changes in coolant flow rate and

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reactor power it has been possible to evaluate the performance of the delayed neutron coolant monitors. With sealed fuel elements, it would be possible to detect the exposure to the coolant stream of a few square cm of enriched fuel. During shutdown for fuel replacement the neutron flux was monitored and variations were related to changes in the core loading. In one carefully conducted series of readings criticality after reloading was predicted to within 0.5%.

B. Kinetics

In the second part of the report the reactivity feedback calculations and measurements are described for charges 3 and 4. Reference is made to the oscillatory and steady state measurements of feedback and to transient response following ramp reactivity change. It is shown that the total amount of fast acting feedback is small after significant burn-up, because of the loss of the fuel axial expansion contribution. The measured responses on a slow time scale are in poor agreement with calculation mainly because the flow and temperature distributions are uncertain in the blanket region. However the kinetic behaviour is reasonably well understood in the region of most interest.

It is concluded that the oscillator measurements (together with the other feedback measurements) provide an extremely valuable guide to the understanding of the kinetic behaviour, and have revealed changes in the characteristics of the D.F.R. core which might not have been found otherwise.



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Abstract

**REACTOR PHYSICS STUDIES FOR THE STEAM GENERATING
HEAVY WATER REACTORS**

BY

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D. Hicks

I. Johnstone

D. C. Leslie

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The special reactor physics problems arising from the use of a boiling steam-water mixture as the coolant in a pressure tube heavy water moderated reactor are discussed with particular reference to the SGHW prototype reactor now under construction at Winfrith. This reactor concept has led to the development of the 5 neutron-group METHUSELAH code for the assessment of core performance for design purposes, and to the THULE scheme of reactor physics calculation, which makes use of the best physical model of the lattice cell currently available.

The phase of the experimental work described concerns the detailed study of the neutron balance in a range of lattices of the pressure tube heavy water type. The work has concentrated on the study of the void coefficient of reactivity and has required the simulation of light water coolant over a range of densities in unpressurised assemblies. Experimental techniques for characterising events in the fast, resonance and thermal neutron energy regions are described. The high accuracy demanded of these techniques is fixed by the requirement that the void coefficient, which is a differential effect, shall be measured with acceptable

accuracy. In addition to these detailed lattice cell measurements, a range of buckling measurements in lattices with and without light water coolant have been used to provide a check on the void coefficient of reactivity and of the effects of leakage on the neutron balance. The use of two sub-critical assemblies and of the DIMPLE critical reactor for these studies is described.

Particular attention has been given to the comparisons of experimental results with the THULE and METHUSELAH predictions and a method for inferring errors in the theoretical predictions, by deducing a value of the infinite multiplication constant from the detailed lattice cell measurements, is presented.

Detailed comparisons of the experimental results from the range of lattices with the latest THULE and METHUSELAH predictions are given with comments on the accuracy of these methods of calculation for detailed design and reactor assessment purposes.



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Abstract

**BRADWELL NUCLEAR POWER STATION - COMPARISON OF SHIELDING PERFORMANCE
WITH DESIGN CALCULATIONS**

by

* F. P. Youell

ø A. K. McCracken

* P. Oliver

1. The results of radiation measurements through the shields of the Bradwell reactors are used to test methods of calculation used in power reactor shield design. The methods used in the U.K., based on a combination of removal and age-diffusion theory, have been developed by the U.K.A.E.A. (Winfrith) Shielding Group and tested using slab shields of simple configuration. However, to treat the effects of complex geometry found in the shields of large power reactors, various extensions to these methods of calculation are necessary and it is the purpose of this paper to test the modified methods by comparing the results of calculation with experimental measurements.

2. Measurements of thermal neutron reaction rates were made using manganese-nickel alloy foils. Fast neutron reaction rates were measured with pellets and blocks of sulphur and gamma dose rates with condenser ion chambers. Results are reported for measurements of:

- (a) fast and thermal neutron reaction rates through the side and top reflectors and pressure vessel,
- (b) fast and thermal neutron reaction rates and gamma dose rates through the side and top biological shields,
- (c) thermal neutron reaction rates in the thermal column

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ø U.K. Atomic Energy Authority, A.E.E. Winfrith Shielding Group
stationed at Harwell

3. The modified methods of calculation for the side and top Bradwell shields are described in detail, particular attention being paid to the treatment of shield layers which do not approximate to uniform slabs of material. An example of this is provided by the top shield where the reflector and chargepans are pierced by the fuel channels, enabling fission neutrons to stream directly from the fuel elements into the pressure vessel and top biological shield.

4. The measured and calculated reaction rates agree to within a factor of 2, over attenuations of 7 or 8 decades. Thus it is shown that the methods of calculation developed for systems with simple geometry may be extended so as to apply to the complex shield configurations of large power reactors.



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Abstract

PHYSICS PROBLEMS OF WATER REACTORS MODERATED BY A
VARIABLE D_2O/H_2O MIXTURE

by

ø F. deWaegh

* J. Smith

Changes in reactivity resulting from burnup of a reactor core can be controlled by the method of variable moderation, using changing mixtures of D_2O and H_2O . This paper reports the physics investigations made in a study of small reactors operating on the pressurised water principle in which the application to marine propulsion has received some special attention. The description centres round the experimental core of 40 MW nominal heat output which is to be installed for test purposes in the BR3 reactor at Mol, Belgium.

The requirements imposed on the core design were that (i) it should have a long operating life, (ii) the parameters of the experimental core should be chosen so that in a power reactor core of the same basic design the shut-off rod capacity would be sufficient to maintain subcriticality even if the reactor were flooded with cold light water at the start of life, (iii) the temperature coefficient of the moderator/coolant should be sufficient to allow regulation by temperature variations over the load range from zero to full power. The paper describes the principles of the physics design which has been evolved, in which the core is built up of an array of standard size hexagonal units, some of which

ø Belgonucleaire, Brussels

* U.K. Atomic Energy Authority, A.E.E., Winfrith

contain fuel and the remainder of which are tubes filled with moderator into which the circular flux-trap type absorber rods are inserted. The methods of calculation for reactivity estimation and burnup, reactivity coefficients, shut-off rod worths and power peaking effects are described. The basis on which the selection of the proportions of moderator to fuel, and coolant to fuel were made are commented on and the relation of power peaking to power output and peak/average burnup discussed. The range of moderator compositions of interest, and the variation with core life is reported.

The core studies have been supported by an initial experimental physics programme at Winfrith, England, and a zero-energy programme at Mol, Belgium, is at present in progress. The two subcritical assemblies and the critical system DIMPLe used in the Winfrith work, and the techniques for the measurement of power peaking, buckling, reaction rates and shut-off rod worths are described and results reported. The analysis of these results by methods developed for the purpose is given and a comparison is made with the design methods used. The VENUS critical assembly at Mol is described and the objects of its programme are given.

Finally a brief description is given of the measurements it is intended to carry out in the BR3 experiment to obtain nuclear and thermal data to aid future designs.



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Abstract

A STUDY OF FUEL CYCLES AND FUEL CYCLE PROBLEMS FOR
ADVANCED GAS COOLED GRAPHITE MODERATED REACTORS

by

* B. Cutts

1. The Advanced Gas-Cooled Graphite Moderated Reactors, like the Magnox Reactors already under construction, will have facilities for fuel replacement on-load. The advantage of this facility apart from the maintenance of a high load factor is that the average core properties in each region of the reactor may be maintained sensibly constant; since the addition of new fuel is counter-balanced by the removal of fully burnt fuel and the changes in fuel properties generally with irradiation. Because the average properties of each region of the reactor are maintained constant the macroscopic flux and power distributions will also remain unchanged during irradiation though the power generated by each piece of fuel (and therefore each channel) will vary as burn-up proceeds.

2. As fuel is irradiated the reactivity it is able to provide in the reactor decreases because of the general burn-out of the fissile elements, the accumulation of fission product poisons and the formation of ^{240}Pu and ^{242}Pu . However, because each region contains fuel at all irradiation levels it is possible to balance the excess neutron production of fresh fuel against the deficiencies of fuel which is almost fully burnt-up. The feed enrichments are therefore reduced well below what would be required if the reactor operated as a whole batch and of course the average irradiation of the fuel approaches closely the maximum that the fuel can stand from an endurance point of view.

* U.K. Atomic Energy Authority, Reactor Group, Risley.

The feed enrichments for a reference design of Advanced Gas-Cooled Reactor are given as follows.

<u>Enriched Uranium fuels</u>		
Critical enrichment	% ^{235}U	1.35
Feed enrichment for 12000 MWD/Te	% ^{235}U	1.68
Feed enrichment for 20000 MWD/Te	% ^{235}U	2.05
<u>Natural uranium and magnox plutonium</u>		
Critical fuel	Kg/Te Pu	3.3
Feed for 12000 MWD/Te	Kg/Te Pu	10.9
Feed for 20000 MWD/Te	Kg/Te Pu	18.5

3. The properties of the reactor in the equilibrium condition are easily established. However, disturbances from this condition will always exist because it will be necessary to change the reactor from a fresh-fuelled state at the start of life to the equilibrium state. This process will take several years. Furthermore improvements in fuel endurance during reactor life and changes in the cost of fissile materials call for a change in the target irradiation or even of the fissile elements used. The effects of such changes on the fuel utilisation and on the ratio of peak to average channel power output need to be known at an early stage since they have some influence on the design of the reactor.



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Abstract

**TEMPERATURE LIMITS AND THE OPTIMISATION OF POWER
OUTPUT IN CALDER AND CHAPELCROSS REACTORS**

- * H.A. Hughes
- * D.S. Briggs
- * R. Dodds
- * R.M. Horsley

The operating limits which are specified for various items of plant determine the performance of a power reactor. These limits are designed to ensure safe and satisfactory operation under normal operating conditions and to minimise the possible hazard arising in the event of a fault. The accumulation of data obtained from the reactor during operation reduces the initial uncertainties in fundamental parameters and actual operating conditions for which due allowance must be made in setting the original limits. Thus improvements in reactor performance can be made.

Temperature is one of the most important limiting parameters in determining the performance of fuel elements, moderator and pressure vessel. In addition to the requirement that there should be no impairment of satisfactory behaviour of fuel during irradiation under normal operating conditions, a limitation is imposed by the criterion that there should be a reasonable probability of avoiding the ignition of single channels in the event of the worst credible fault, a depressurisation incident. The calculations and assessment of experimental data which enable the uncertainty of prediction of the complete steady-state temperature distribution to be determined, the calculation of transient rises suffered by each channel, and the application of these calculations and other data to the determination of operating limits are described.

* U.K. Atomic Energy Authority, Production Group, Windscale

Limitations on the extent of irradiation damage to graphite, phenomena which are also temperature dependent, can be defined from both operational and safety considerations. The main properties discussed in the paper are growth, stored energy and air-oxidation, the first two being limited by a minimum graphite temperature and the third by a maximum temperature. Growth is monitored by changes in Wigner gap size, the measurements being correlated with dimensional changes produced by irradiating specimens in high flux reactors after correcting the latter results for differences in temperature and dose-rate. A limiting temperature is then set by the criterion that no operational difficulties should arise during the life of the reactor. Stored energy limits are dictated by the criterion that a single channel fire should not spread to other channels. Oxidation of graphite introduces a further heat source into a depressurisation fault when air has been drawn into the reactor, and a maximum temperature is set which takes into account the operative measures which it is reasonable to suppose can be carried out if such an incident occurs.

The temperature limits on the pressure vessel are fixed by necessity to avoid the occurrence of two main phenomena. The first, which governs the minimum temperature, is brittle fracture, and is studied by out-of-pile and in-pile specimen measurements. The second phenomenon is stress rupture, which sets an upper temperature limit so that the time to rupture at the stress levels obtaining in the vessel, making due allowance for relaxation due to creep, will exceed the required life. The results of analysis of stress rupture data from specimens, and the application of this to the particular conditions are described.



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Abstract

CONTROL, STABILITY & PROTECTION OF A GAS-COOLED, GRAPHITE-MODERATED REACTOR

by

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A.B. Whiteley

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Central Electricity Generating Board, London.

The paper is limited to control, flux stability, and protection against reactivity faults in large CO₂-cooled, graphite-moderated reactors of the type now in operation or under construction in Great Britain. The steady state design limitations are stated, derived from limits set by fault criteria in relation to fuel element properties. The difficulties of correctly flattening the radial flux shape of such a reactor are discussed, with examples.

The assumptions made in deriving the necessary control rod investment to shut down a reactor under all conditions of temperature and irradiation are examined and it is shown that, for maximum economy in investment, the control rod pattern must be carefully chosen. The effect of locally removing control rods from a fully shut-down reactor is illustrated.

Reactor flux stability is examined, both in the radial/azimuthal and axial directions, and it is concluded that some form of control is necessary to prevent the former, but not the latter, possible instabilities. The use of lightly absorbing control rods for this purpose is described, using a typical large reactor as an example. Although such rods have been provided against instabilities, it has been found convenient to incorporate them into an overall reactor control scheme. Faults in the control system of these rods can, however, lead to locally high temperatures, and a full and detailed examination is made of this type of fault condition within the reactor.

The problems associated with reactor start-up are discussed and the development of multi-group bulk control rod lifts is traced starting from the

single group life as used at Calder Hall. It is shown that axial flux distortion is thereby reduced, but that, unless the control rod pattern is carefully chosen, radial/azimuthal flux distortions can give local high temperatures in the fuel elements. Symmetry of core layout is emphasised as a design concept.

Bulk reactivity faults, caused by all rods moving simultaneously, are shown to be very severe, although their severity depends upon the relative disposition of the groups of rods when the reactor passes through criticality; the necessity to provide interlocks to high safety standards to prevent the occurrence of such faults is demonstrated. It is shown, nevertheless, that fast start-ups can be achieved on this type of reactor provided certain basic design requirements can be economically met.

Part-load operation of a reactor is discussed, and methods of overriding the xenon poison following a load reduction are described.

The recognition of the existence of many of the problems discussed, has been due to the development of improved methods of calculation using large digital and analogue computing machines. Using the former, a method of calculation is described which enables reactor transient conditions in 3 dimensions to be examined, and another one-dimensional method for transient calculations is also presented; these have been used in producing the curves shown in this Paper.



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Abstract

DESIGN OF CONTROL SYSTEMS FOR GAS-COOLED REACTORS

by

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A. J. Spurgin

United Power Co. Ltd., London.

The practical application of control principles to gas-cooled, graphite moderated reactors is discussed, with particular reference to the grouping, spatial distributions and movement rates of controlling absorbers. Both static and kinetic aspects of control are considered; some requirements can only be expressed qualitatively, and some are conflicting, necessitating compromise; adequate monitoring of reactor conditions is essential.

Speedy and economical design analysis methods are necessary to complement fully detailed analysis methods, which are more suitable for detailed operational studies of reactor behaviour. Typical design methods suitable for static and kinetic control analysis are described.

Various groupings of the coarse control rods for start-up are considered and it is shown that a four group scheme probably provides the best arrangement. The use of automatic spatial control rods for both unison and local control is described, the opposing demands affecting movement rates are discussed and an (x,y) alternative layout to the conventional sector scheme is introduced. The merits of fixed and moveable absorbers for power flattening are compared; the need to consider the spatial control and override rod groups as integral parts of the flattening and the importance of symmetry in the rod groups are emphasised.

The control principles discussed above are illustrated with reference to two contrasting systems. Firstly, the Tokai design is considered; this is a

relatively small reactor at an advanced stage of construction with many features common to other Stage 1 Magnox reactors. The design includes secondary shut-down devices using boron balls to give increased protection against earthquake hazards. The way in which these devices, as well as the control groups and variable flattening system, are accommodated within a single grid of standpipes is described, particular attention being given to eliminating the possibility of "local criticality". Typical spatial transient behaviour for the sector arrangement of controllers is illustrated and the effects of control thermocouple location relative to the rods are indicated.

Secondly a new core design is considered and the control system described; this core is much larger than that of the Tokai reactor and fixed flattening is utilised. Because of the large size it is possible to extend the integration of spatial control and override rods with the flattening; this improves both the core efficiency and the automatic flattening effect of the spatial control rods. The advantages of an (x,y) arrangement of spatial control and override rods for a large core are discussed. Results of radial-azimuthal stability studies and transients following local absorber movement are given.

The likely trends in control systems for future reactors, such as the advanced gas-cooled reactor, are discussed, and include closer integration with conventional plant, better load following, and increasingly flexible operation.



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Abstract

**DEVELOPMENT OF LICENSING AND INSPECTION OF
NUCLEAR INSTALLATIONS IN THE UNITED KINGDOM**

***T. Griffiths**

***R. Gausden**

The control of major nuclear installations, excluding those operated by the United Kingdom Atomic Energy Authority, is regulated by the Nuclear Installations (Licensing and Insurance) Act, 1959. The Act requires an operator of a nuclear installation to hold a licence and sets out the licensee's liability and the provision he must make for covering it. It is a requirement that there shall be attached to the licence conditions as may be necessary or desirable in the interests of safety.

The Act specifies that a site shall not be used for installing or operating a nuclear reactor without a licence and gives powers for the prescription by regulation of other types of nuclear installations within a broad definition.

The administration of the Act requires that it shall be determined what installations should be prescribed and what conditions in the interests of safety should be attached to each licence.

The paper deals with the principles which have been applied and indicates how these were met in the original drafting of the regulations and conditions. Comments are made on the experience of the first few years of applying the regulations and conditions and outlines some of the changes which were made as the result of that experience. The paper also refers to the development of the siting policy for Nuclear Reactors and of the organisation of the Inspectorate which was set up for the administration of the Act.

*Ministry of Power, London



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Abstract

**REACTOR SAFETY ANALYSIS AS RELATED
TO REACTOR SITING**

by

***F. R. Farmer**

One of the many techniques employed in reactor safety assessment is to postulate a certain series of hypothetical faults, to study their consequences and in particular to determine whether a fission product release occurs and, if so, of what magnitude.

One such hypothetical accident is sometimes identified as the 'maximum credible' and the consequent fission product release, expressed as a single number, then acquires exaggerated importance and conceals the uncertainties both in the definition and the analysis.

The use of a routine pattern of reactor fault analysis establishes excessive familiarity with such concepts as the 'burst duct accident' and containment which then take on an unjustifiably precise meaning.

The present paper illustrates the inadequacy of this approach, explores the difficulties and suggests the need for maintaining a critical approach to fault analysis and flexibility in guides used for reactor siting and licensing.

* U.K. Atomic Energy Authority, Authority Health and Safety Branch, Risley.



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Abstract

THE SAFETY OF U.K. GAS COOLED REACTORS - A REVIEW

by

* G. R. Bainbridge Z J. R. Harrison
+ L. F. Bowles / J. M. Yellowlees
= L. Cave

During the period 1954 - 1964 development of the gas cooled graphite moderated reactor has been advanced in the U.K. to the stage where electricity generation is practical, with economic and safety standards comparable with other non-nuclear systems.

This paper outlines the progressive development of the design of the U.K. nuclear stations, and its effect on their safety. At the beginning of the decade only the Harwell and Windscale single-pass air cooled piles were operating, but by the time of the 1958 Geneva Conference several closed circuit CO₂ cooled reactors were in operation at Calder Hall, with magnox clad uranium metal fuel.

The valuable design and operating experience from them, and the many civil nuclear power stations since constructed, has been fed into the logical development of the Advanced Gas-Cooled Reactor, with stainless steel clad uranium oxide fuel. The Windscale experimental A.G.R. has operated at power for more than a year and the first large civil station of this type is being designed. From the design and operating experience which has accumulated, worthwhile site relaxations could now be allowed for new nuclear stations in the U.K.

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- * U.K. Atomic Energy Authority, Reactor Group, Risley
 - + English Electric Co., Whetstone, near Leicester
 - = United Power Co., London
 - Z Central Electricity Generating Board, London
 - / The Nuclear Power Group, Knutsford

For each station to date extensive safety studies have been done, including fault analyses of the kind outlined in this paper. There has been close co-operation between the Consortia, the Electricity Boards, the Licensing Authorities and the U.K.A.E.A. to analyse and understand the general and special safety features of each station and to summarise the results of this analysis in the safety reports for individual stations. This study of safety problems, from the time of initial site selection to the time of full power operation, will continue for each new station on a scale which its individual features demand.

There have been many improvements made in the designs, in the understanding of the equipment required for emergencies and in operating methods, as a result of safety analysis, and this paper gives an account of typical examples.

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Abstract

EXPERIENCE IN THE TRANSPORT OF IRRADIATED FUEL

by

- * N. F. C. Bishop
- * J. C. Egan
- * W. W. Marshall

1. The routine movements of irradiated fuel within the United Kingdom are described; this includes the transport of irradiated magnox canned natural uranium fuels by road from Chapelcross to Windscale and by rail from the U.K. civil nuclear power stations to Windscale. Details are provided of the transport containers, the vehicles and the general organization of these movements. Also included is a description of a container in which dry magnox canned elements are transported.
2. Similar information is given for the transport of M.T.R. type fuel elements both between Harwell and Dounreay and from Australia and Denmark to Dounreay.
3. The extension of this transport service to further overseas reactors is discussed with particular reference to the technical and engineering support work in aid of the transport of irradiated magnox fuel from Italy and Japan to reprocessing plants in the U.K. This includes a description of the type of vehicle which could be used both for transporting magnox fuels in water at low temperature to minimise corrosion effects and also fuels from advanced gas cooled and water moderated reactors where mechanical cooling equipment is required to maintain regulatory temperature requirements.
4. An outline of the supporting design and development work is given; this includes a description of the experimental work on the transport of magnox clad fuels in aqueous heat transfer media which led to the decision to use mechanically cooled transport containers. The selection and use of alternative heat transfer media is discussed and some indication is given of the problems which arise from the use of organic media.

* U.K. Atomic Energy Authority, Production Group, Risley

5. Indications of cost are provided for transport containers and vehicles and of freight rates for both rail and sea journeys.

6. It is shown that as the design of transport equipment is closely related to regulatory requirements substantial reductions in the capital cost of this equipment may not be possible but that reductions in the overall cost of transport operations could be achieved by maintaining high equipment utilisation factors. It is suggested that the current trend in which reactor operators accept responsibility for the return of irradiated fuel is uneconomic and that, the growth of a limited number of specialist transport organisations should be encouraged.

to the mean value of about 3×10^{-6} gm/gm.hr. at 400°C . Samples trepanned from the moderator of operational reactors and specimens exposed in reactors have indicated that the reaction rate could increase by at least an order of magnitude after prolonged exposure to reactor conditions, and it is possible to correlate the increase with irradiation dose and with the weight loss which results from exposure to the reactor coolant. In other experimental work which is described the performance of chemicals as inhibitors of graphite oxidation has been investigated, and one, phosphorus oxychloride, shows promise of reducing the reaction rate by at least an order of magnitude if injected in small quantities at the time of an emergency.

The influence of some of the above factors on the course of events following the postulated accident is illustrated using calculated temperature - time curves for a reactor channel, and the methods of calculation are outlined. The effect of graphite oxidation on reactor temperatures is strongly dependent on the reaction rate which is assumed. The value for virgin material, 3×10^{-6} gm/gm.hr as measured at 400°C , has a negligible effect, although values very much greater than this could lead to rising graphite temperatures. The remedial measures provided for such an emergency include a massive CO_2 feed, and it is shown that this, combined with circulator flow at atmospheric pressure, would prevent damage to fuel and release of fission products following such an accident. Chemical inhibitors if used would provide a valuable additional safeguard.



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Abstract

CONTROL OF GAS-BORNE ACTIVITY ARISING FROM REACTOR FAULTS

R. D. COLLINS x

A. E. J. EGGLETON +

To ensure that any gas-borne activity arising from an accident to a reactor of the A.G.R. type will be adequately trapped, the preferred trapping system (which consists of a charcoal adsorber and high efficiency filters) has been tested over the relevant range of conditions. I-131 is the most dangerous fission product and the difficulty in trapping it is related to the chemical and physical forms which it can assume. The trapping system was therefore tested with fission products released from irradiated fuel pencils heated in various CO₂/CO/air mixtures. At temperatures a little above the melting point of the can, releases of iodine, tellurium and caesium were between 10 and 50% but that of strontium was less than 0.2%.

It was found that the principal features of the behaviour of the I-131 in the release could, by a suitable technique, be usefully reproduced using separated I-131. A considerable proportion was carried on fume, but no penetration of the absolute filter was detected. Vapour compounds were found which did not deposit rapidly on copper surfaces and the most penetrating of them were identified as methyl and other alkyl iodides, previously found in separated carrier-free iodine diluted with clean air. Interconversion of the various forms was studied in the gas phase and on surfaces.

x U.K. Atomic Energy Authority, T.R.G., Windscale

+ U.K. Atomic Energy Authority, A.E.R.E., Harwell

Charcoal was found to delay methyl iodide by physical adsorption but only to retain it permanently if the loading was below a certain limit. For the standard material (chosen for its good retention of other forms) the limit was about $1\mu\text{g/g}$ of charcoal at 100°C , and less at room temperature but alternative charcoals have been found with a higher limit even at room temperature.

Freshly released fission products at concentrations of $10^{-3} - 10^{+3} \mu\text{g/m}^3$ were found to penetrate a trapping system including a 10 cm. charcoal bed to an extent ranging from one part in 10^3 at the lower concentration to one in 10^5 at the higher, and methyl iodide formed a considerable proportion of the gaseous activity.

The only serious penetration of any fission product occurred in an experiment in which large amounts of activity were released in conditions favouring compound formation. Prolonged elution of the heavily loaded bed resulted in penetration by iodine activity identified mainly as alkyl iodides. The total loading was an order of magnitude above the methyl iodide limit for the charcoal used, but lay within the limit of satisfactory operation for some charcoals since tested.



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Abstract

TECHNIQUES AND ECONOMICS OF RADIOACTIVE WASTE TREATMENT

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B. F. WARNER *
D. W. CLELLAND *

An outline is given of the systems which have been used at the U.K.A.E.A., Windscale site in Cumberland for the treatment of radioactive liquid wastes arising from two major reprocessing plants.

The source, quantity, activity and composition of the various waste arisings at Windscale are given and a comparison is made of the wastes arising from the first and second reprocessing plants. The arisings and treatment systems are categorised for description and discussion according to their activities as "highly active", "medium active" or "low active".

In general the effectiveness of the various methods which have been employed are discussed and the main considerations which contribute to safety and economy are emphasised. The main improvements in the waste systems which have recently been made during the construction of the second reprocessing plant at this site are indicated.

A section on highly active wastes describes the existing facilities at Windscale for their concentration and storage, mentions the main limiting features of the present plant, points out the main problems associated with the more highly active wastes which will arise from the U.K. Civil Power Programme and indicates present plans for dealing with future wastes of this type.

Sections of the paper dealing with medium and low active wastes describes the systems which have been operated at Windscale and indicate how recent changes have made the systems more effective and economical.

Progressive development of waste handling techniques at Windscale has, without reducing safety standards, resulted in increased efficiency and greater economy in operation.

* U.K. Atomic Energy Authority, Production Group, Windscale



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Abstract

WORK IN THE UNITED KINGDOM ON FIXATION
OF HIGHLY RADIOACTIVE WASTES IN GLASS

by

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*J. R. Grover

*W. H. Hardwick

The paper describes the present state of development of a process for the conversion of radioactive waste liquors from a plant processing civil reactor fuel to a glassy solid suitable for long term storage. Glasses based on the borosilicate system fulfil the requirements for the Harwell process but phosphate glasses appear to offer some important advantages. The pilot plant experiments to date have been conducted with borosilicate glass. The results of sixteen runs on the active pilot plant, FINGAL, are discussed. Decontamination factors from feed to condensate of 10^8 are achieved for isotopes such as ^{90}Sr and ^{137}Cs . The same decontamination factor is also achieved for long periods for Ruthenium, but ultimate breakthrough lowers the overall value for Ru to 10^5 . Ruthenium trapped on a filter is not volatilized again when the filter is incorporated in the glass of a subsequent cycle. Only one part in 10^{13} of the total activity was detected in the exhaust gas from the process when 1,000 curies of mixed fission products were vitrified during 42 hours operation.

The industrial application of the FINGAL process is considered for the fixation of the highly active waste arising from a 6,000 MW magnox reactor power programme involving 2,000 tonnes of uranium per year. The number of disposable cylinders required, the number of fixation units required, and the method of storage are assessed for various possible operating conditions.

Data show how rates of conversion of liquid to glass and temperatures generated in storage vary with size of disposable process vessel. It is

* U.K. Atomic Energy Authority, Research Group, Harwell

concluded that liquid wastes should be stored for two years and then solidified in vessels twelve inches in diameter. Estimates have shown that the process costs are comparable with those for storage of highly concentrated liquid in tanks. These costs represent less than 1% of the value of electricity generated from the fuel.